NON-PUBLIC?: N
ACCESSION #: 9111050247
LICENSEE EVENT REPORT (LER)

FACILITY NAME: COMANCHE PEAK - UNIT 1 PAGE: 1 OF 07

DOCKET NUMBER: 05000445

TITLE: REACTOR TRIP RESULTING FROM ERRATIC OPERATION OF THE MAIN TURBINE
ELECTROHYDRAULIC CONTROLLER
EVENT DATE: 10/03/91 LER #: 91-023-00 REPORT DATE: 11/04/91

OTHER FACILITIES INVOLVED: N/A DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 030

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: D. E. BUSCHBAUM COMPLIANCE TELEPHONE: (817) 897-5851
SUPERVISOR

COMPONENT FAILURE DESCRIPTION:
CAUSE: SYSTEM: COMPONENT: MANUFACTURER:
REPORTABLE NPRDS:

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On October 3, 1991, Comanche Peak Steam Electric Station Unit 1 was in Mode 1, Power Operation, with the reactor at approximately 30 percent of rated thermal power. In order to collect data for evaluation of a problem with the main turbine electrohydraulic control (EHC) system, the idle hydraulic fluid pump was started and one of the two operating pumps was secured. Following the pump switchover, erratic operation of the main turbine steam control valves created a pressure pulse which propagated up the main steam line to the steam generator, causing a spike of sufficient magnitude on the steam generator narrow range level transmitter to exceed the Hi-Hi level setpoint. The main feedwater pumps tripped automatically, and the reactor was manually tripped due to decreasing steam generator levels. Causes include erratic operation of the EHC system and sensitivity of the steam generator narrow range level

instrumentation. Corrective actions for EHC system performance anomalies will be determined based on the results of evaluation of test data.

END OF ABSTRACT

I. DESCRIPTION OF THE REPORTABLE EVENT

A. REPORTABLE EVENT CLASSIFICATION

An event or condition that resulted in a manual or automatic actuation of any Engineered Safety Feature (ESF) including the Reactor Protection System.

B. PLANT OPERATING CONDITIONS PRIOR TO THE EVENT

On October 3, 1991, just prior to the event, Comanche Peak Steam Electric Station (CPSES) Unit 1 was in Mode 1, Power Operation, with reactor power at 30 percent.

C. STATUS OF STRUCTURES, SYSTEMS, OR COMPONENTS THAT WERE INOPERABLE AT THE START OF THE EVENT AND THAT CONTRIBUTED TO THE EVENT

There were no inoperable structures, systems or components that contributed to the event.

D. NARRATIVE SUMMARY OF THE EVENT, INCLUDING DATES AND APPROXIMATE
TIMES

On July 13, 1991, a reactor trip occurred while troubleshooting a seal problem on one of the three hydraulic fluid pumps (EIIS:(P)(JJ)) in the main turbine electrohydraulic control (EHC) system (EIIS:(JJ)) (the event is described in LER 91-020). Evaluation of data recorded during the event and operating experience with the EHC system led to the conclusion that the event was caused by unstable control fluid flow following control fluid pump switchover, leading to rapid and erratic movement of the main turbine steam control valves (EIIS:(V)(JJ)). The rapid closure of the turbine control valves created a pressure pulse which propagated up the main steam line to the steam generator (EIIS:(SG)(SB)), causing a spike of sufficient magnitude on the steam generator narrow range level transmitter (EIIS:(LT)(JB)) to exceed the Hi-Hi level setpoint. The resulting turbine trip initiated a reactor trip.

During the evaluation of the event, it was decided that additional testing was required to gather data for further evaluation of the design and operational characteristics of the EHC system. To provide useful data, the testing was to be performed with the EHC system in the load control mode, which required that turbine power be above approximately 20 percent.

On October 3, 1991, Comanche Peak Unit 1 was in Mode 1, Power Operation. A power reduction was in progress in preparation for entering the first refueling outage later that day. Because of the recognized fisk associated with testing the EHC system, a management decision was made to perform testing at the lowest practical power level prior to entering the outage.

At approximately 1715 CDT the power reduction had been halted with reactor power at 30 percent and turbine power at approximately 24 percent to allow testing of the . EHC system. Hydraulic fluid Pump B was started and Pump A was secured, leaving Pumps A and C in operation. At 1719 a Hi-Hi level signal was received on Steam Generator 2 resulting in a turbine trip and automatic trip of the operating main feedwater pump (EIIS:(P)(SJ)). Because reactor power was below 50 percent, the turbine trip did not result in an automatic reactor trip. At 1723 the Shift Supervisor (utility, licensed) ordered a manual reactor trip due to decreasing steam generator levels.

Control Room personnel (utility, licensed) responded in accordance with emergency operating procedures, and by approximately 1800 the plant was stabilized in Mode 3, Hot Standby. At 1947, the NRC was notified of the event via the Emergency Notification System in accordance with 10CFR50.72.

E. THE METHOD OF DISCOVERY OF EACH COMPONENT OR SYSTEM FAILURE, OR
PROCEDURAL OR PERSONNEL ERROR

During performance of the EHC system test, Control Room personnel were monitoring
he various plant parameters which
could be expected to indicate a recurrence of the EHC system operating anomaly observed on July 13. Numerous alarms were

received in the Control Room indicating apparent level
fluctuations in the steam generators leading eventually to a
turbine trip.

II. COMPONENT OR SYSTEM FAILURES

A. FAILURE MODE, MECHANISM, AND EFFECT OF EACH FAILED COMPONENT

There have been no failed components identified as having
contributed to this event.

B. CAUSE OF EACH COMPONENT OR SYSTEM FAILURE

No failed components have been identified.

C. SYSTEMS OR SECONDARY FUNCTIONS THAT WERE AFFECTED BY FAILURE OF
COMPONENTS WITH MULTIPLE FUNCTIONS

No failed components have been identified.

D. FAILED COMPONENT INFORMATION

No failed components have been identified.

III. ANALYSIS OF THE EVENT

A. SAFETY SYSTEM RESPONSES THAT OCCURRED

As a result of the Hi-Hi steam generator level signal, a
turbine trip signal, steam dump signal, and feedwater isolation
signal were generated; those functions actuated and all
associated components performed as designed. The Auxiliary
Feedwater system (EIIS:(BA)) started automatically due to the
loss of main feedwater flow.

B. DURATION OF SAFETY SYSTEM TRAIN INOPERABILITY

There were no safety systems or components rendered inoperable
during or as a result of the event.

C. SAFETY CONSEQUENCES AND IMPLICATIONS OF THE EVENT

The turbine trip event leading to a reactor trip is discussed in Section 15.2.3 of the CPSES Final Safety Analysis Report. The analysis uses conservative assumptions to demonstrate that Departure from Nucleate Boiling Ratio will never decrease below the limiting value of 1.30 during the event. The event of October 3, 1991, occurred at 30 percent reactor power, and all protective functions responded as required. The event is completely bounded by the FSAR accident analysis which assumes an initial power level of 102 percent and makes conservative assumptions which reduce the capability of safety systems to mitigate the consequences of the transient. It is concluded that the event of October 3 did not adversely affect the safe operation of CPSES Unit 1 or the health and safety of the public.

IV. CAUSE OF THE EVENT

IMMEDIATE CAUSE

The immediate cause of the event was erratic operation of the main turbine electrohydraulic control system following pump switchover. This condition resulted in rapid movement of the turbine control valves which in turn generated pressure pulses which exceeded the Hi-Hi steam generator level setpoint. Decreasing steam generator levels resulting from the automatic trip of the operating main feedwater pump led to the decision to manually trip the reactor.

ROOT CAUSES

The root cause of the event is considered to be a malfunction of the EHC system which resulted in unstable control fluid flow following control fluid pump switchover. The nature of the malfunction is considered to be related to one of the two following conditions:

1) The entrainment of air bubbles in the fluid flow following the introduction of air into a static portion of the system by an as yet unidentified mechanism resulted in minor variations in EHC system response as the entrained air bubbles swept past the controlling edge of the pilot valves;

2) Oscillations in control fluid pressures and flows as the operating pumps attempted to achieve steady state dynamic flow

conditions following pump switchover resulted in variations in EHC system response.

The variations in EHC system response caused the erratic operation of the main turbine control valves and the resultant pressure pulses sensed by the steam generator narrow range level instrumentation.

An additional cause of the event has been determined to be the sensitivity of the steam generator narrow range level instrumentation to pressure pulses of very shod duration but relatively high magnitude, such as those generated during rapid movement of the main turbine steam control valves. These pressure waves propagate upstream to the water filled impulse lines of the steam flow transmitters and steam generator narrow range level transmitters. The short duration signals were not an actual indication of steam generator inventory, but were responsible for the automatic trip of the operating main feedwater pump which precipitated the decision to manually trip the reactor.

V. CORRECTIVE ACTIONS

Cause: Malfunction of the EHC system

Corrective Action: Additional EHC system testing will be performed during the current refueling outage. The results of that testing along with data collected previously will be evaluated to determine the appropriate action required to allow pump switchover without recurrence of the event.

Cause: Sensitivity of steam generator level instrumentation

Corrective Action: A design modification has been initiated to install a filter card with a lag time constant in the steam generator narrow range level instrumentation channels. The modification will minimize unnecessary ESF actuations from Hi-Hi and Lo-Lo level signals brought about by pressure spikes. Completion of the modification is expected prior to completion of the current refueling outage.

VI. PREVIOUS SIMILAR EVENTS

LER 91-020 describes a reactor trip resulting from erratic operation of the main turbine EHC system. The diagnostic testing of the EHC system described in LER 91-020 was in progress at the time of the

October 3 event, and the modification to the steam generator narrow range level instrumentation had not yet been implemented.

ATTACHMENT 1 TO 9111050247 PAGE 1 OF 3

Figure "Correspondence Sign-Off" omitted.

ATTACHMENT 1 TO 9111050247 PAGE 2 OF 3

Log # TXX-91362
File # 10200
10CFR 50.73(a)(2)(iv)
TUELECTRIC

November 1, 1991
William J. Cahill, Jr.
Group Vice President

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D. C. 20555

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION
DOCKET NO. 50-445
ENGINEERED SAFETY FEATURE ACTUATION
LICENSEE EVENT REPORT 91-023-00

Gentlemen:

Enclosed is Licensee Event Report 91-23-00 for Comanche Peak Steam Electric Station - Unit 1, "Reactor Trip Resulting from Erratic Operation of the Main Turbine Electro Hydraulic Controller."

Sincerely,

William J. Cahill, Jr.

OB/ds

c - Mr. R. D. Martin, Region IV
Resident Inspectors, CPSES (2)

400 N. Olive Street L.B. 81 Dallas, Texas 75201

ATTACHMENT 1 TO 9111050247 PAGE 3 OF 3

TUELECTRIC

Austin B. Scott, Jr.
Vice President
CPSES-9127945
October 31, 1991

No Response Required

TO: W. J. Cahill - E22

SUBJECT: LICENSEE EVENT REPORT 50-445/91-023-00
REACTOR TRIP RESULTING FROM ERRATIC OPERATION OF THE MAIN TURBINE ELECTROHYDRAULIC CONTROLLER

Attached is Licensee Event Report (LER) 50-445/91-023-00, which has been prepared in accordance with 10CFR50.73(d). This LER has been reviewed by SORC (Meeting No. 91-090), and recommended for approval. Additionally, I have reviewed and approved the LER and find it acceptable for submittal to the NRC (required by November 4, 1991).

If you should have any questions, please contact D. E. Buschbaum at extension 5851.

A. B. Scott, Jr. O10

GGD:jcc

Attachment

cc: CCS E06
R. D. Walker ST-24

*** END OF DOCUMENT ***